

DOCUMENT-IDENTIFIER: US 4659511 A  
TITLE: Method for solidifying radioactive waste

DEPR:

As still another embodiment of this invention, there will be described below, with reference to FIG. 12, a case where not the pelletized radioactive waste but the radioactive waste (mainly composed of Na.sub.2 SO.sub.4) generated from a nuclear power station is directly treated and solidified in a 200-l drum. In this case, in order to secure the strength of the solidified waste and the proper volume reduction ratio of the waste, the radioactive liquid waste contained in a tank 12 is first dehydrated and formed into a powder in a dryer 13 and then supplied into a tank 14. Various methods are known for drying the radioactive liquid waste, such as centrifugal film drying, spray drying, fluidized bed drying, drum drying, freeze drying and crystallization, and any of these methods can be employed in this invention.

CCOR:  
588/4

CCXR:  
588/14

DOCUMENT-IDENTIFIER: US 4980093 A

TITLE: Method of treating high-level radioactive waste liquid

DEPR:

As described above, in accordance with the present invention, the high-level radioactive waste liquid can be separated into (a) condensate (nitric acid solution), (b) bulk wastes mainly containing sodium nitrate, sodium hydroxide, etc., and (c) residue (nitrate) mainly containing fission products, actinoids, and corrosion products (iron, chromium, nickel, etc.) in the reprocessing process. As a result, (1) since the high-level radioactive waste liquid is separated by the freeze-vacuum-drying process, (2) the problem of corrosion due to high temperature is overcome, (3) and the condensate (nitric acid solution) and bulk wastes mainly containing sodium nitrate and sodium hydroxide can be converted into low-level radioactive waste. In addition, (4) if the residue containing fission products, actinoids, and corrosion products (iron, chromium, nickel, etc.) that have stable configurations as nitrate and roasted product is stored in the form of residue, it is advantageous at the time when useful elements (rhodium, palladium, etc.) contained in the residue (nitrate or roasted material) are recovered in the future. Furthermore, (5) when a glassified body is made, since the amount of sodium is small, a high-quality glassified body can be formed, so that the amount of a glassified body produced can be reduced substantially. Moreover, fractionated substances can be used as raw material for artificial minerals (e.g., titanium, zirconium, calcium, barium, aluminum) .

CCOR:

588/20

DOCUMENT-IDENTIFIER: US 3849330 A  
TITLE: CONTINUOUS PROCESS FOR IMMOBILIZING RADIONUCLIDES,  
INCLUDING CESIUM AND  
RUTHENIUM FISSION PRODUCTS

DEPR:

Advantageously, ruthenium fission products which are highly volatile and are not immobilized in the complex metalo-silicate as are the other radio nuclides, excepting of course cesium, will be reduced to ruthenium metal by either elemental silicon or elemental aluminum and be separated with the other metals, e.g., iron, rhodium, palladium, nickel, chromium, silicon via container 19. When container 19 is filled, the temperature of the silicate melt is lowered by cooling coils 20, freezing the silicate and container 19 consisting of a metal ingot 21 containing noble metals (Ru, Rd, Pd) is disconnected from reactor 1 and an empty container is attached in its place.

CCOR:

588/14

DOCUMENT-IDENTIFIER: US 4115311 A

TITLE: Nuclear waste storage container with metal matrix

TTL:

Nuclear waste storage container with metal matrix

BSPR:

Nuclear waste storage containers containing a metal matrix for the nuclear

waste are desired to provide greater impact strength for the waste container and to increase the thermal conductivity to prevent undesirably high centerline temperatures. Matrix fabrication has been effected by casting, although this process may prohibit the use of materials having high melting points which may be required to prevent melting of the matrix during storage from unforeseen temperature excursions or other accidents. Matrix fabrication by prior processes has generally been concerned with low temperature metals such as lead and aluminum.

BSPR:

It is a further object of this invention to provide a storage container containing high-level nuclear waste in a sintered metal matrix, which metal matrix provides impact strength to the waste container and increases heat transfer from the container interior to the container exterior to prevent undesirably high centerline temperatures within the container.

DEPR:

The high-level waste particles that may be used in this invention may be obtained from glass particle fabrication processes or ceramic particle fabrication processes wherein the waste form utilizes existing calcining processes with a compositionally modified waste liquid to achieve an improved or supercalcine waste form in which generally all of the radioactive atoms will be isolated in thermally and chemically stable phases. For

example, the high level waste particles may be chemical vapor deposition alumina and pyrolytic carbon coated improved ceramic particles or supercalcine ceramic particles, and generally contain fission products as ceramic oxides or as glass modifiers.

The metal matrix powders that may be used are such as pure copper and its alloys, pure iron and its alloys, AISI Series 410, 304, 310 and 316 stainless steels, and superalloys, such as Inconel or Hastalloy. A particularly good copper alloy matrix is manganese bronze having a nominal composition, expressed in weight percent, of 57.5 copper, 39.25 zinc, 1.25 iron, 1.25 aluminum, and 0.25 manganese. The properties of the matrix materials that are desirable are a high melting point, good thermoconductivity, good mechanical strength, good corrosion characteristics in salt, water, and/or air, and good oxidation resistance in air at operating temperatures.

CCOR:

588/15

DOCUMENT-IDENTIFIER: US 5980602 A  
TITLE: Metal matrix composite

BSPR:

It is still an additional object of the present invention to provide a method for manufacturing an improved metal matrix composite material to result in a material having superior hardness, strength, and density characteristics while being extrudable and weldable for use in the manufacture of a variety of structural members which may be readily connected to one another such as in bicycle and other vehicle frames and components, engine components, aircraft parts, tooling, sporting equipment such as tennis rackets, badminton rackets, baseball bats, arrows, golf club shafts, and hockey and lacrosse sticks, eyewear, automotive parts, electronic parts, furniture, medical equipment, battery housings, nuclear shielding, marine components, robots, carts and seats, gourmet cookware, toy casings, high-pressure containers, tank linings, and armor, for example.

DEPR:

3. A metal matrix composite of aluminum alloy 6061 base metal material and 30 weight % boron carbide. This composite material is extrudable and exhibits a tensile strength of 62.3 kpsi and a yield strength of 58.4 kpsi. The formulation may be used for structural stiffness for marine applications or nuclear shielding since it has strength and corrosion resistance.

CCOR:

75/236

CCXR:

75/249

DOCUMENT-IDENTIFIER: US 5700962 A  
TITLE: Metal matrix compositions for neutron shielding applications

BSPR:

According to an aspect of the present invention, a neutron shield is made of a boron carbide-metal matrix composite wherein the metal matrix material is aluminum, magnesium, titanium, or gadolinium, or an alloy thereof. The composite is formed by blending dry powders of boron carbide and the metal matrix material to uniformly mix the powders, and then subjecting the powders to high pressures to transform the powders into a solid body that is then sintered to form a composite that can be extruded, cast, forged, welded, and manufactured into structures for neutron shielding. Such structures include containers for holding nuclear waste, and load-bearing plates for use in neutron shielding structures in nuclear submarines and power plants.

CCOR:  
75/236

DOCUMENT-IDENTIFIER: US 4814046 A

TITLE: Process to separate transuranic elements from nuclear waste

BSPR:

The current proposed process for treating the waste IFR salt does not recover

the contained actinides, but converts the wastes into more readily disposable

forms. The waste salt is contacted with a cadmium-lithium alloy, a strong

reductant, to transfer nearly all of the actinides from the salt into the metal

phase. This also results in most of the rare earth fission products being

transferred into the metal phase. The treated salt is dispersed in a cement

matrix that is cast into corrosion-resistant metal containers.

This waste is

highly radioactive because it contains fission product cesium and strontium,

but it may not require disposal in a deep geologic repository because it does

not contain significant amounts of transuranic elements. The cadmium-lithium

alloy that contains the actinides and rare earths extracted from the salt is

combined with other metal wastes. The mixture is retorted to vaporize the

cadmium and leave a metallic residue consisting of fission products, small

amounts of actinides, zirconium from the fuel alloy and fuel cladding hulls.

This residue is combined with a metal powder, such as copper, and pressed into

a solid ingot. The metal matrix is encapsulated in a corrosion resistant

container and, because it contains small, but significant amounts of TRU

elements, it must be buried in a geologic repository.

CCXR:

75/398



DOCUMENT-IDENTIFIER: US 4488990 A

TITLE: Synthetic monazite coated nuclear waste containing glass

BSPR:

Due to corrosion problems with the glass produced by this process, an alternative process known as the cermet process, has also been developed. In the cermet process an iron nickel base metal matrix contains the fission product oxides from the high level waste. The cermet produced by this process is a continuous electrically conductive metal matrix containing small particles (about 1 micron) of waste oxide. The process is described more fully in U.S. Pat. No. 4,072,501. The final sintered waste cermet articles produced by this process could be in the form of pellets or rods. Currently cylindrical pellets 0.6 inches in diameter and 0.6 inches long are conveniently extruded.

CCOR:

588/11

CCXR:

588/15

DOCUMENT-IDENTIFIER: US 4383944 A

TITLE: Method for producing molded bodies containing highly active radioactive wastes from glass granules embedded in a metallic matrix

BSPR:

By proper selection of the glass-to-metal ratio, the size of the individual granules or of the powder particles and the mixing conditions, a product is obtained in which the glass granules or the glass powder containing the radioactive waste fission products are discontinuously embedded in a continuous metal matrix phase. The average interparticle space ( $\lambda$ ) for the glass granules is given by  $\lambda = L \cdot V_M^{-1/3}$  where  $L$  is the average size (intercept length) of glass granules and  $V_M$  is the volume content of the metal phase. The equation provides the interrelationship between  $\lambda$ ,  $L$  and  $V_M$  to keep a proper distance ( $\lambda \geq L$ ) between the discontinuously embedded glass granules. The time of mixing depends on the sort of powders. Generally good distributions of the glass granules in the metal matrices which is to say discontinuity of the glass particles--are obtained the more the condition is approached that  $\lambda = L$  where  $\rho$  means density. If there should exist access to a granule or a powder particle from outside the molded body by corrosion or leaching media, only this one granule or particle is in contact with the environment, while all others remain insulated. Substantial corrosion and leaching is thereby prevented.

CCOR:

588/11

CCXR:

588/15

DOCUMENT-IDENTIFIER: US 4383855 A  
TITLE: Cermets and method for making same

DEPR:

As briefly pointed out above, the cermets of the present invention may incorporate radioactive waste in quantities up to from 1 to about 75 wt.% without deleteriously affecting the cermet quality. In making radioactive waste cermets, compositional adjustments to the urea heavy-metal bearing admixture may be necessary to ensure the formation of a desired metal matrix and ceramic phases to provide the cermet with high density, thermal conductivity, mechanical integrity, self-shielding properties, and resistance to corrosion or leaching. Further, compositional adjustments may include the addition of iron, nickel, copper, molybdenum, tin, lead, and the like for providing sufficient oxide for the liquid-phase sintering operation to assure that the particulate material, including nonreduced oxides, metals, and other impurities that are present in radioactive waste, is microencapsulated within the cermet. It may be also desirable to adjust the ceramic portion of the cermet to include additives such as the oxides of zirconium, titanium, aluminum, silicon, phosphorus, manganese, barium, and calcium. These oxides are precursors to the formation at elevated temperatures of known fission product immobilizing and retentive phases, such as: hollandite, perovskite, zirconolite, nepheline, monazite, fluorite, spinel, leucite, pollucite, scheelite, opatite, and kalsilite. See, for example, Ringwood, A. E., et al, "The SYNROC Process: A Geochemical Approach To Nuclear Waste Immobilization," Geochem. J., 13, pp. 141-165 (1979); and Boatner, L. A., et al, "Monazite and Other Lanthanide Orthophosphates as Alternate Actinide Waste Forms," The

Scientific Basis for Nuclear Waste Management, Vol. II, C. J. Northrup, Ed., Plenum Press, New York, New York (1980).

DEPR:

In order to further illustrate and demonstrate the effectiveness of the present invention, a typical manufacturing operation is set forth below. A simulated thorex waste solution, such as obtained from Nuclear Fuel Services-type thorex waste, was prepared by admixing iron, nickel, copper, cobalt, and thorium in a dilute nitric acid. This admixture was concentrated by sufficient heating to drive off water. This solution was then contacted with molten urea and precipitated at above 180.degree. C. The recovered precipitate was then calcined at 800.degree. C. to provide an admixture of various calcined powders. These powders were mixed with a water binder and cold pressed into a green pellet of a size in a range of about 10.times.10 mm. This pellet was placed in a furnace under hydrogen atmosphere and heated from room temperature to a temperature of about 1500.degree. K. over a fifteen minute period. During this heating, the selected oxide mixture in the pellet was sufficiently reduced to a substoichiometric or suboxide value to provide the liquid phase necessary for sintering the remaining particulate material when the temperature increased to a level greater than the eutectic temperature of approximately 1300.degree. K. The sintered pellet was removed from the furnace and cooled to room temperature in a reducing atmosphere. The resulting article was examined and found to be structurally sound and shock resistant. Metallographic examination revealed that all oxide particles were tightly encapsulated within a uniformly dispersed continuous metal matrix.

CCXR:

75/951

DOCUMENT-IDENTIFIER: US 4338215 A

TITLE: Conversion of radioactive wastes to stable form for disposal

ABPL:

Radioactive waste material, such as that resulting from radioactive weapons plant operation or from nuclear fuel reprocessing, in suitable form, such as radionuclide-containing oxide and/or oxyhydroxide and/or hydroxide particles, is held by a metal or metal "alloy" to an electrically conductive cathodic material upon which the metal or alloy is electrodeposited. In this way the radionuclide species including strontium and/or cesium, which are biologically extremely hazardous, are incorporated into a metal matrix held to a base and may be disposed of, as by underground storage, in such form, which is considered to be more resistant to dissolution by ground water and to damage by mechanical stresses arising from tectonic activity than are glasses or ceramics incorporating radionuclides. In improvements of the process and of the resulting radionuclide-including article, the article is electrolytically or otherwise covered or coated with suitably corrosion resistant and mechanical damage resistant covering(s) or coating(s), as by continuing electrodeposition of the metal, sometimes after addition of more material, which is a source of the metal, to a suitable electrolyte, or after replacement of the electrolyte.

BSPR:

In accordance with the present invention there is provided a method for converting radioactive waste material into a stable article for disposal or storage which comprises electrolyzing a bath containing ions of a corrosion-resistant, electrodepositable metal or of a plurality of materials including at least one such metal, in the presence of a solid state radioactive

waste material which contains a radionuclide selected from the group consisting of strontium 90 and cesium 137 and mixtures thereof in one or more oxide and/or oxyhydroxide and/or hydroxide forms so that the metal or materials including metal is/are deposited on a cathode and bind(s) the radioactive waste thereto. Preferably, the radioactive waste treated is calcined weapons waste or wastes from nuclear fuel reprocessing, the cathode and the matrix metal are of copper or other similarly corrosion resistant and electrodepositable metal, and the electrolyte dissolves little, if any, of the radionuclide compound(s) of the waste, e.g., 5% or less, preferably less than 0.5 or 0.1%. Also, it is preferred that the article resulting from the process described be covered or overcoated, either electrolytically or otherwise, with a corrosion resistant metal or other suitable coating, as by continuing electrodeposition of the matrix metal after consumption of all or substantially all of the radioactive waste material, or by electrodeposition of covering metal or alloy from a new or replenished bath. While the plating metal may be codeposited with finely divided particles of radionuclide compounds, it is sometimes preferable to pelletize the radionuclide compounds resulting from treatment of the waste and then hold them to a conductive substrate by simultaneously metal coating them and the substrate. In such operations, and when fine particles of radionuclide compound are being electrodeposited, it may be desirable to treat the pellets or powders first, as by coating or mixing with conductive powder, to increase surface electrical conductivity and to reduce solubility. The described process may be continuous or of the batch type. Also within the invention are the described articles and a method for disposal of radioactive waste materials in the form of such an article .

DEPR:

In FIG. 5 the step represented by numeral 39 is the conversion to a sludge or slurry of an aqueous nuclear waste solution or suspension. This is usually effected by evaporation of the continuous medium and the product resulting may contain insoluble particles of radionuclide(s), preferably in oxide, oxyhydroxide and/or hydroxide forms. However, such conversion to a sludge or slurry may be by other means, including sorption by zeolites and the utilization of thickening or solidifying agents or other materials, which may promote subsequent drying of the waste solution or obtaining of desirable forms thereof after drying. In some cases the nuclear waste suspension may be evaporated directly, despite being relatively dilute, but usually for best evaporation a thickened liquid or sludge will be preferred as a starting material. Drying or calcining of the sludge, usually to oxide, oxyhydroxide or hydroxide form, preferably to the oxide, is effected at step 41 and, although it is not shown, the dried product may be further size reduced, agglomerated or pelletized for best dispersion and/or most desirable settling and codeposition rates and characteristics, e.g., reduced solubility in the electrolyte. Increasing particle size helps to diminish solubility and so is often preferable. Before or after such size changing operation (or in the absence of it) it is preferred that to promote codeposition the conductivity of the normally poorly conductive or even insulating oxides will be increased, as by electroless deposition thereon of conductive material, intimate mixing with a finely divided conductor, spray treatment with metal or other application of a conductive substance or substances to the dried particles. Often, metals are preferred as the conductive materials, but graphite, especially colloidal graphite, has also been found to be very useful. After

subjection of the particles to a suitable conductivity increasing operation 43 (which may sometimes be omitted), the produce thereof (if finely divided) is dispersed in a metal salt solution (usually an aqueous electrolyte) for codeposition of the radionuclide-containing compound and metal onto a metal base. The making of the metal salt solution is represented at 45 and the solution is shown mixed with and utilized with the radionuclide-containing compound in electrodeposition step 47. In FIG. 4 coating onto one major surface of a conductive cathode is shown and in codeposition step 47 such coating is effected and additionally, coating of the other major surface of the cathode may be carried out after a 180.degree. rotation thereof subsequent to depositing of a sufficient thickness of metal matrix-radionuclide-containing compound on the first major cathode surface. Step 49 of FIG. 5 represents a subsequent electrolytic coating of the article with a suitable protective metal or binary metal alloy (which may or may not be the same as the matrix material) to provide additional corrosion resistance. While such step is highly preferred, it is within the present invention to utilize the uncoated article of operation 47, and it is also within the invention to utilize such product when it is coated on only a single surface (usually a major surface) of the cathodic base material. After completion of the coating operation, represented in step 49 (after 47 if subsequent coating is not used) the article is stored or disposed of, as at 51, preferably being ultimately disposed of in safe underground storage. The process described is also readily adaptable to electrolytic (and electroless) deposition of pellets or comparatively large bodies of rad-waste onto base materials.



DOCUMENT-IDENTIFIER: US 4209421 A  
TITLE: Method of preparing bodies containing radioactive substances

BSPR:

For the safe final disposal of highly radioactive wastes, it is known to add glass formers to them and, by known methods, to melt a glass therefrom which, after solidification, can be stored in the form of monolithic glass blocks in an appropriate container. Furthermore, products of extra safety margin have been developed, such as aggregates of glass and metal, in which the highly radio-active glass in the form of particles, which may be of sizes between two and eight millimeters, is embedded in a metal matrix (W. Heimerl, Atomwirtschaft-Atomtechnik, 20 (1975) pp. 347-349). In other methods, the highly radioactive glass block of appropriate composition is subjected to a controlled devitrification by a suitable heat treatment; examples of the kind of glass ceramics which have been proposed for formation by this method are those of the celsian, perowskite, diopside and eucryptite type (A.K. De, B. Luckscheiter, W. Lutze, G. Malow, E. Schwiewer, S. Tymochowicz, Management of Radioactive Wastes from the Nuclear Fuel Cycle, IAEA, Vienna 1976, Vol. II, pp. 63-73).

CCOR:  
588/15

CCXR:  
588/11

DOCUMENT-IDENTIFIER: US 4204975 A

TITLE: Method and apparatus for encapsulating radioactively contaminated lumps or granular material in metal

DEPR:

The apparatus for carrying out the method of the invention can be used to advantage for encapsulating spent nuclear fuel particles of the kind used in high-temperature nuclear reactors. These particles consist of a core of uranium carbide that was encased in a three-fold coating of pyrolytic carbon, silicon carbide and pyrolytic carbon. The diameter of the uranium carbide cores thus coated was between about 400 and 500 .mu.m. The spent nuclear fuel particles encased, according to the process of the invention, with aluminum as the matrix metal, were suitable for final storage after the riser pipe 9 and the air-removal pipe 8 had been pinched off. Instead of pinching off the tube ends, it is of course also possible to close them in some other way, for example by welding.

CCOR:

588/15

DOCUMENT-IDENTIFIER: US 4072501 A

TITLE: Method of producing homogeneous mixed metal oxides and metal-metal oxide mixtures

DEPR:

The simulated fission products and nickel were dissolved as nitrates in a minimum amount of 1 M HNO<sub>3</sub>, about 180 ml. Sufficient solid urea was added to provide about 20 grams of oxides per liter of urea. The solution was heated to 130.degree.-140.degree. C for 1 hour to remove the water of dissolution and hydration, after which the temperature was raised to about 180.degree. C for 40-50 minutes to precipitate the metal values. All the metal values precipitated homogeneously. The molten urea mixture was heated to 400.degree. C for sufficient time to evaporate the urea leaving a dry powder. This dried powder was heated to 800.degree. C in air for calcining, forming a mixture of nickel oxide and simulated fission and corrosion product oxides. The dried powder was heated at 850.degree. C for 1 hour in hydrogen atmosphere, whereupon the nickel oxide was reduced to metal powder. The metal values from simulated fission products were still present as oxides having typical particle size of 0.2 microns. The powders were homogeneously mixed. The mixture was consolidated by hot pressing at 1100.degree. C and 4000 psi for 1 hour to form a cermet waste immobilization article containing fission products within a continuous nickel matrix. Scanning electron microscopy revealed that each oxide particle was completely surrounded by the metal matrix. For waste immobilization articles, the matrix metal can be 50 to 70 volume % of the article. If desired, a ceramic waste immobilization article can be prepared by eliminating the reduction step and hot pressing the oxide powder mixture.

Cermet waste immobilization articles are substantially more ductile and thermally conductive than ceramic waste articles. The present process is particularly advantageous for preparing radioactive powders and for other applications requiring remote handling, since the solids are separated from solution by evaporation rather than filtration. Filtration processes are generally to be avoided in handling hazardous substances, e.g., nuclear fuel reprocessing operations, in order to preclude the formation of aqueous effluents.

CCOR:  
75/344

CCXR:  
75/365

DOCUMENT-IDENTIFIER: US 5879110 A

TITLE: Methods for encapsulating buried waste in situ with molten wax

BSPR:

In the early days of the nuclear age, contaminated debris and undocumented low level radioactive waste were buried in shallow trenches. Other waste materials were placed in underground storage tanks. These burial areas are now considered to pose a unacceptable risk to the environment. Excavation and removal of these wastes is potentially dangerous and very expensive. The concern is that excavation of such sites could release airborne radioactive contaminants which would pose a substantial harm to personnel and nearby residents. There have been a number of solutions proposed for containing these sites. Some of these solutions include slant drilled jet grouting, soil freezing, soil dehydration, tunneling, and chemical grout permeation. Others have taught vertical drilling and hydraulic fracturing as a means of forming a bottom barrier.

CCXR:

588/249

DOCUMENT-IDENTIFIER: US 5905184 A

TITLE: In situ construction of containment vault under a  
radioactive or  
hazardous waste site

BSPR:

In the early days of the nuclear age, contaminated debris and undocumented low level radioactive waste were buried in shallow trenches. Other waste materials were placed in underground storage tanks. These burial areas are now considered to pose a unacceptable risk to the environment. Excavation and removal of these wastes is potentially dangerous and very expensive. The concern is that excavation of such sites could release airborne radioactive contaminants which would pose a substantial harm to personnel and nearby residents. There have been a number of solutions proposed for containing these sites. Some of these solutions include slant drilled jet grouting, soil freezing, soil dehydration, tunneling, and chemical grout permeation. Others have taught vertical drilling and hydraulic fracturing as a means of forming a bottom barrier.

CCOR:

588/260

CCXR:

588/249

DOCUMENT-IDENTIFIER: US 5890840 A

TITLE: In situ construction of containment vault under a radioactive or hazardous waste site

BSPR:

In the early days of the nuclear age, contaminated debris and undocumented low level radioactive waste were buried in shallow trenches. Other waste materials were placed in underground storage tanks. These burial areas are now considered to pose a unacceptable risk to the environment. Excavation and removal of these wastes is potentially dangerous and very expensive. The concern is that excavation of such sites could release airborne radioactive contaminants which would pose a substantial harm to personnel and nearby residents. There have been a number of solutions proposed for containing these sites. Some of these solutions include slant drilled jet grouting, soil freezing, soil dehydration, tunneling, and chemical grout permeation. Others have taught vertical drilling and hydraulic fracturing as a means of forming a bottom barrier.

CCXR:

588/249